VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10 CFR 50.54(f)

January 31, 2003

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No. 02-689
NL&OS/GDM R2
Docket Nos. 50-280/281
50-338/339
License Nos. DPR-32/37
NPF-4/7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA AND SURRY POWER STATIONS UNITS 1 AND 2
NRC BULLETIN 2002-01 - REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY
REQUEST FOR ADDITIONAL INFORMATION

On March 18, 2002, the Nuclear Regulatory Commission (NRC) issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The bulletin required all holders of operating licenses for pressurized water reactors to provide, within 60 days, the basis for concluding that their boric acid inspection programs provide reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and within the bulletin. If a documented basis did not exist, licensees were requested to provide their plans, if any, for a review of their programs. Virginia Electric and Power Company (Dominion) provided a description of its boric acid corrosion control programs for Surry and North Anna Power Stations in a letter dated May 16, 2002 (Serial No. 02-168A).

In a subsequent request for additional information dated November 21, 2002, the NRC requested North Anna and Surry Power Stations to address six additional questions associated with their boric acid corrosion control programs. Dominion's response to these six questions is provided in the attachment.

As discussed in our 60-day response letter noted above, the boric acid corrosion control (BACC) programs implemented at North Anna and Surry Power Stations have been and continue to be effective at detecting and correcting boric acid leakage that could cause wastage or degradation of the reactor coolant system pressure boundary. These programs ensure compliance with applicable regulatory requirements. However, in light of recent industry issues, the BACC programs for North Anna and Surry are being combined into a common program and enhanced to provide further assurance of

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the prompt identification and correction of potential boric acid leakage. The enhanced BACC program (e.g., inspection frequencies, techniques and locations) will be evaluated and periodically revised as necessary to reflect lessons learned and industry operating experience.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

L. N. Hartz
Vice President - Nuclear Engineering

Attachment

Commitments made in this letter are as follows:

1. Inspections of inaccessible locations based on PWSCC susceptibility and risk significance, utilizing guidance published by EPRI and the Westinghouse Owners Group. The scope of scheduled inspections will be expanded to include ASME Class 1, 2 and 3 bolted connections, as well as Alloy 600 tubing and Alloy 82/182 weldments. ASME Class 1 bolted connections and Alloy 600 tubing and Alloy 82/182 weldments will be inspected every refueling outage. Bare-metal visual (BMV) exams of locations highly susceptible to PWSCC will be performed every RFO. The ASME Class 2 and 3 bolted connections will be inspected once each inservice inspection period or on a schedule determined by engineering evaluation, EPRI/WOG recommendations, and/or operating experience. BMV exams locations with lower PWSCC susceptible, such as the bottom mounted instrumentation (BMI) tubing will be performed on a frequency based on EPRI/WOG recommendations. North Anna and Surry Units 1 and 2 RPV BMI Alloy 600 tubes and 82/182 welds will be visually inspected from the exterior surface of the insulation every RFO and underneath the insulation on a frequency based on EPRI/WOG recommendations.

2. We intend to perform a "best effort" inspection of the replacement RPV heads and VHP nozzles consistent with the recommendations provided in NRC Bulletin 2002-02 for plants with less than eight EDY, unless formal industry guidance supports alternate inspection scope and schedules for PWSCC resistant materials. Any change to the North Anna or Surry RPV heads and VHP nozzles inspection scope or schedule will be communicated to the NRC.

3. If evidence of leakage is detected on a bottom mounted instrumentation tube at North Anna or Surry, insulation will be removed to assess the extent of the leak and for potential wastage of the lower RPV head.

4. Bare-metal visual inspections of the Surry steam generator channel head drain nozzles will be performed every refueling outage in accordance with the enhanced BACC program. If leakage is detected, the extent of wastage at the leakage source and components in the leak path will be assessed prior to repair.

5. If evidence of boric acid leakage is found in inaccessible areas, the insulation will be removed for a detailed follow-up assessment to determine the extent of the leak and to determine if any degradation exists. Alternatively, an engineering assessment could be performed to justify either leaving the insulation in place or deferring the removal of insulation until a later date.

cc: U. S. Nuclear Regulatory Commission
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Serial No: 02-689
Docket Nos.: 50-280/281
50-338/339
Subject: RAI – NRC Bulletin 2002-01

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 31st day of January, 2003.

My Commission Expires: May 31, 2006.

Vicki L. Hull
Notary Public

(SEAL)

Attachment

Request for Additional Information
NRC Bulletin 2002-01

North Anna and Surry Power Stations Units 1 and 2

Virginia Electric and Power Company
(Dominion)

Request for Additional Information
NRC Bulletin 2002-01

North Anna and Surry Power Stations Units 1 and 2

General Overview

In a letter dated November 21, 2002, the NRC requested additional information regarding the boric acid corrosion control (BACC) programs implemented at North Anna and Surry Power Stations. Dominion initially implemented its BACC programs in 1989 to address leakage/wastage/degradation concerns expressed in NRC Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The BACC programs for North Anna and Surry are being combined into a single, common program, and the scope of the BACC program is being expanded to address additional NRC concerns set forth in NRC Bulletin (BL) 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," and in BL 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," as well as in consideration of recent industry events and operating experience associated with boric acid corrosion.

The goal of the BACC Program is to prevent boric acid related degradation of the Reactor Coolant System (RCS) and associated ASME Class 1, 2 or 3 components, as well as, any related degradation to intervening structures, systems or components (SSCs). To achieve that goal, the BACC program ensures that borated water leakage is consistently identified, documented, evaluated, trended, effectively repaired and monitored to prevent subsequent leakage. Specifically, the program requirements include the following:

- Determination of principal locations where coolant leaks smaller than allowable Technical Specification limits could cause degradation of the RCS pressure boundary by boric acid corrosion.

- Establishment of visual examination procedures to locate small coolant leaks, establishing both the source of the leak and its target (i.e., where the boric acid travels) in order to determine the extent of degradation.

- Methods to conduct and document examinations to identify boric acid deposits identified during walkdowns or during system pressure tests.

- Corrective actions to prevent recurrences of this type of leakage, including trending and use of utility operating experience and industry working group recommendations.

- Methods to perform Engineering evaluations of boric acid leakage and/or material degradation associated with boric acid leakage.

The scope of the BACC program encompasses inspection, operational, and maintenance activities, which identify borated water leaks, including:

- Programmatic visual examinations and pressure testing of the RCS and ASME Class 1, 2 and 3 borated water systems,
- Walkdowns by engineering, operations, ISI/NDE or maintenance personnel, which can occur during containment entries at power, during plant shutdowns and outages, and during plant startups, and
- On-line leakage monitoring of the RCS pressure boundary by the Control Room Operators.

Response to NRC Questions

1. *Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.*

   Dominion Response:

   The insulation used on the RCS at North Anna and Surry is predominantly formed stainless steel reflective removable insulation. This insulation is designed to be used in segments around the component and is held together with spring clips and stainless steel lock wire. Station specifications for the installation of insulation also permit the use of mineral wool, fiberglass, and calcium silicate. The majority of the RCS insulated components are located in the loop rooms in containment approximately 15 to 20 feet off the floor. The elevation of this equipment adds to the personnel safety concern already created by performing these inspections while the RCS is at elevated temperatures. Attached Tables 1 and 2 for North Anna and Surry, respectively, provide listings of Unit 1 components that are currently periodically inspected for boric acid leakage and degradation of threaded fasteners by ISI Engineering personnel. Unit 2 components are similar. Operations/Engineering personnel also perform walkdowns of RCS pressure boundary components/zones to inspect for boric acid accumulation during unit cooldown and startup.

Currently, insulation is removed for each ASME XI Class 1 bolted connection to permit visual inspections every refueling outage (RFO). Similarly, insulation is removed from Class 2 bolted connections when visual inspections are performed every inservice inspection period (once every two or three RFOs). There are presently no requirements to remove insulation from ASME Class 3 bolted connections to perform visual examinations. However, if evidence of boric acid leakage is identified on the insulation of any Class 3 bolted connection, the insulation would be removed to identify the source of the leakage and whether any degradation exists.

Dissimilar metal Alloy 82/182 weld locations are currently examined under insulation by non-visual nondestructive evaluation (NDE) ultrasonic (UT) techniques every inspection interval (once per ten years). Presently, there are no NDE requirements to examine small diameter welds (i.e., below three-inch pipe diameter).

Walkdowns of Class 1, 2, and 3 borated piping systems are currently performed for boric acid leakage without removal of insulation during "at power" containment entries, plant shutdowns and outages, as well as during plant startups. Insulation is removed if a boric acid trail disappears underneath insulation or if boric acid deposits mask a susceptible component. Insulation may also be removed if discoloration, staining, boric acid residue or other evidence of component leakage is identified.

In light of recent industry operating experience, the current GL 88-05 BACC program is being enhanced to address additional concerns regarding primary water stress corrosion cracking (PWSCC) and wastage. The enhanced program will establish inspection locations and frequencies based on PWSCC susceptibility and potential for wastage, as recommended by Westinghouse and EPRI. The scope of scheduled inspections will be expanded to include ASME Class 1, 2 and 3 bolted connections, as well as Alloy 600 tubing and Alloy 82/182 weldments. ASME Class 1 bolted connections and Alloy 600 tubing and Alloy 82/182 weldments will be inspected every refueling outage. Bare-metal visual (BMV) exams of locations highly susceptible to PWSCC will be performed every RFO. The ASME Class 2 and 3 bolted connections will be inspected once each inservice inspection period or on a schedule determined by engineering evaluation, EPRI/WOG recommendations, and/or operating experience. BMV exams locations with lower PWSCC susceptible, such as the bottom mounted instrumentation (BMI) tubing, will be performed on a frequency based on EPRI/WOG recommendations. North Anna and Surry Units 1 and 2 RPV BMI Alloy 600 tubes and 82/182 welds will be visually inspected from the exterior surface of the insulation every RFO and underneath the insulation on a frequency based on EPRI/WOG recommendations. The following Alloy 600 BMI tubes and Alloy 82/182 weld locations will be inspected every RFO:

- Pressurizer surge, spray, safety and relief nozzle Alloy 82/182 butter welds at North Anna 1 and 2 (Surry 1 and 2 pressurizers do not have Alloy 82/182 welds),

- Steam generator (SG) primary loop nozzle Alloy 82/182 butter welds (North Anna 1 only. North Anna 2 and Surry 1 and 2 do not have such welds exposed to borated water), and

- SG channel head drain nozzles (Alloy 600 tubes and 82/182 welds) at Surry 1 and 2 (North Anna 1 and 2 have internal channel head drain nozzles that will not leak to the outside if cracked).

North Anna and Surry Units 1 and 2 RPV bottom mounted instrumentation (BMI) Alloy 600 BMI tubes and 82/182 welds will be visually inspected from the exterior surface of the insulation every RFO and underneath the insulation on a frequency based on EPRI/WOG recommendations.

Since the Alloy 600 reactor pressure vessel (RPV) head control rod drive mechanisms (CRDMs) and Alloy 82/182 J-groove welds at North Anna and Surry Power Stations are classified as highly susceptible to PWSCC cracking, replacement RPV heads have been ordered. The North Anna Unit 2 reactor vessel head has been replaced, and replacement RPV heads for North Anna Unit 1 and Surry Units 1 and 2 are currently scheduled for installation during the next RFO for each unit in 2003. The vessel head penetration (VHP) nozzles and the attachment welds in the replacement RPV heads are constructed of materials (Alloy 690 CRDM tubes and Alloys 52/152 J groove welds) that are inherently much more resistant to PWSCC than Alloy 600 and Alloys 82/182. With the replacement of the RPV heads, North Anna Units 1 and 2 and Surry Units 1 and 2 will be in the least susceptible category for PWSCC [<8 effective degradation years (EDY)]. Based on this recategorization, we intend to perform a "best effort" inspection of the replacement RPV heads and VHP nozzles consistent with the recommendations provided in NRC Bulletin 2002-02 for plants with less than eight EDY, unless formal industry guidance supports alternate inspection scope and schedules for PWSCC resistant materials. Should additional industry information become available to justify an alternative inspection scope or frequency, we intend to reconsider the proposed future inspection plans for North Anna Units 1 and 2 and Surry Units 1 and 2 RPV heads and VHP nozzles. Any change to the North Anna or Surry RPV heads and VHP nozzles inspection scope or schedule will be communicated to the NRC.

2. *Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in* <u>*inaccessible areas.*</u> *In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.*

Dominion Response:

Inaccessible areas are defined as those locations that are blocked from direct view by nearby components, or require scaffolding for access, and/or are situated in a high radiation area. In these cases, an initial boric acid corrosion visual inspection is made of surrounding or low lying areas including the floor or equipment areas under the inaccessible components or other areas where leakage may be channeled. Where possible, the use of mirrors or binoculars is employed. If evidence of boric acid leakage is found in inaccessible areas, the insulation will be removed for a detailed follow-up assessment to determine the extent of the leak and to determine if any degradation exists. Alternatively, an engineering assessment could be performed to justify either leaving the insulation in place or deferring the removal of insulation until a later date. Two existing ASME code cases are also applicable for use in the BACC program as follows:

- N-533-1, which allows examination of a bolted joint with insulation in place as long as a VT-2 exam is performed during the next RFO with the insulation removed, and

- N-566-2, which allows for a sampling of bolts for removal rather than removal of all of the bolts at a leaking flange.

Relief requests have been approved for both North Anna and Surry to use these two code cases or similar relief.

Currently, boric acid leakage walkdowns are performed whenever the unit is shut down, i.e., during scheduled refueling outages and unscheduled maintenance outages. Boric acid leakage walkdowns are also performed during plant startups. "At power" walkdowns are also performed to identify leakage sources/targets whenever the unidentified reactor coolant system leakrate meets specified, conservative action levels over the previously calculated leakrate. At present, insulation is removed from bolted connections and other susceptible locations during scheduled pressure tests, during scheduled ISI pressure boundary inspections, during scheduled inspections of RCS and borated ASME Class 1 and 2 bolted connections, or whenever boric acid deposits found in the vicinity of the component indicate the need for assessing the potential for wastage.

The scope of scheduled inspections will be expanded to include ASME Class 1, 2 and 3 bolted connections, as well as Alloy 600 tubing and Alloy 82/182 weldments. ASME Class 1 bolted connections and Alloy 600 tubing and Alloy 82/182 weldments will be inspected every refueling outage. Bare-metal visual (BMV) exams of locations that are highly susceptible to PWSCC will be performed every RFO. The ASME Class 2 and 3 bolted connections will be inspected once each inservice inspection period or on a schedule determined by engineering evaluation, EPRI/WOG recommendations, and/or operating experience. Bare-metal visual

(BMV) exams of locations with lower susceptible to PWSCC, such as the bottom mounted instrumentation (BMI) tubing, will be performed on a frequency based on EPRI/WOG recommendations.

Leakage detection is determined by RCS leak rate calculations, containment sump level monitors, and containment particulate/gas monitors.

3. *Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.*

Dominion Response:

If through-wall cracks were to exist at these lower head BMI penetrations, boric acid deposits flowing down a leaking thimble guide tube should be readily visible outside the insulation due to gravity. A VT-2 inspection of this area is performed every refueling outage as part of the ISI Program. If evidence of leakage is detected on a BMI tube at North Anna or Surry, insulation will be removed to assess the extent of the leak and for potential wastage of the lower RPV head. No such BMI penetration leakage has been identified to date at North Anna or Surry. Furthermore, BMI tubing leakage due to PWSCC has not been verified at any other PWR worldwide, including Ringhals (plants with same RV fabricators as North Anna and Surry). No measurable wastage has been identified on the bottom head of any RPV worldwide, even those heads which have experienced leaking cavity seals. No other components subject to wastage are located in the vicinity of the BMI tubes at North Anna or Surry.

As noted above, North Anna and Surry Units 1 and 2 RPV Alloy 600 BMI tubes and 82/182 welds will be visually inspected from the exterior surface of the insulation every RFO and underneath the insulation on a frequency based on EPRI/WOG recommendations.

It is furthermore noted that even if a double-ended break of a BMI tube were to develop during operation, a step change in charging flow, containment radiation monitoring equipment and/or the incore sump level alarm would provide an early indication of leakage to the Control Room operators. Performing a controlled shutdown and cooldown of the unit would mitigate this event. Following the shutdown, the remaining leakage would be recovered to the Liquid Waste or Boron

Recovery System, and the RCS level would be maintained as normal via the charging system.

4. *Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.*

   Dominion Response:

   Apart from Alloy 600 BMI nozzles at North Anna and Surry, the only remaining small diameter Alloy 600 nozzles that could lead to wastage if cracked are the steam generator (SG) bottom channel head drain nozzles at Surry 1 and 2. North Anna 1 and 2 have internal SG bottom channel head drain nozzles, which cannot leak boric acid onto external carbon steel surfaces even if a through-wall crack was to exists. Currently, bare-metal visual inspections of the channel head drain nozzles are being performed as a result of industry operating experience, but are not currently part of the GL 88-05 BACC program. However, the inspections of the Surry channel head nozzles will be incorporated into the enhanced BACC program and performed every RFO. If leakage is detected, the extent of wastage at the leakage source and components in the leak path will be assessed prior to repair.

5. *Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.*

   Dominion Response:

   Currently, inspections are being performed in accordance with Dominion's response to Generic Letter 88-05. The enhanced BACC program expands the scope and inspection frequency of the original inspection program based on PWSCC susceptibility and risk significance. The relative PWSCC susceptibility of Alloy 600 locations was calculated utilizing the methodology in MRP-48, Section 2.1.

6. *Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.*

<u>Dominion Response</u>:

Westinghouse reviewed its databases and applicable communications to determine whether Westinghouse had made any recommendations to the owners of Westinghouse nuclear steam supply systems (NSSSs) regarding the need for visual inspections of Alloy 600 and 82/182 materials in the reactor coolant system pressure boundary. The detailed review did not identify any such Westinghouse recommendations.

Table 1

Components Inspected for Boric Acid Leakage
North Anna Power Station Unit 1 (Unit 2 Similar)

I. Visual Inspection of ASME Class 1 RCS Pressure Boundary Components

Component Inspected	Location/Description	Insulated/Not Insulated
Pressurizer Cubicle – Elevation 291 Feet		
1-RC-SV-1551A	Top of PRZR	Insulated
1-RC-SV-1551B	Top of PRZR	Insulated
1-RC-SV-1551C	Top of PRZR	Insulated
1-RC-MOV-1535	Top of PRZR	Insulated
1-RC-MOV-1536	Top of PRZR	Insulated
1-RC-PCV-1455C	Top of PRZR	Insulated
1-RC-PCV-1456	Top of PRZR	Insulated
PRZR manway	1-RC-E-2	Insulated
Pressurizer Cubicle – Elevation 262 Feet		
1-RC-PCV-1455A	Pzr Spray	Insulated
1-RC-PCV-1455B	Pzr Spray	Insulated
1-CH-328	3-inch Aux Spray	Not Insulated
Reactor Coolant Pump "A" Motor Cubicle – Elevation 262 Feet		
1-RC-P-1A	Casing/Seal House Bolting	Not Insulated
1-RC-E-1A	Primary Manway (Hot Leg)	Insulated
1-RC-E-1A	Primary Manway (Cold Leg) Insulated	Insulated
1-RC-MOV-1585	8-inch Bypass MOV	Insulated
Reactor Coolant Pump "A" Motor Cubicle (beneath grating) – Elevation 262 Feet		
1-RC-MOV-1590	Hot Leg MOV	Insulated
1-RC-MOV-1591	Cold Leg MOV	Insulated
1-SI-127	12-inch SI to Tc	Insulated
1-SI-83	6-inch SI to Tc	Insulated
Reactor Coolant Pump "B" Motor Cubicle – Elevation 262 Feet		
1-RC-P-1B	Casing/Seal House Bolting	Not Insulated
1-RC-E-1B	Primary Manway (Hot Leg)	Insulated
1-RC-E-1B	Primary Manway (Cold Leg)	Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
1-RC-MOV-1586	8-inch Bypass	Insulated

	MOV	
Reactor Coolant Pump "B" Motor Cubicle (beneath grating) – Elevation 262 Feet		
1-RC-MOV-1592	Hot Leg MOV	Insulated
1-RC-MOV-1593	Cold Leg MOV	Insulated
1-SI-144	12-inch SI to Tc	Insulated
1-SI-86	6-inch SI to Tc	Insulated
1-CH-325	3-inch CH to Tc	Insulated
1-CH-496	3-inch CH to Tc	Insulated
Reactor Coolant Pump "C" Motor Cubicle – Elevation 262 Feet		
1-RC-P-1C	Casing/ Seal Housing	Not Insulated
1-RC-E-1C	Primary Manway (Hot Leg)	Insulated
1-RC-E-1C	Primary Manway (Cold Leg)	Insulated
1-RC-MOV-1587	8-inch bypass MOV	Insulated
Reactor Coolant Pump "C" Motor Cubicle (beneath grating) – Elevation 262 Feet		
1-RC-MOV-1594	Hot Leg MOV	Insulated
1-RC-MOV-1595	Cold Leg MOV	Insulated
1-SI-161	12-inch SI to Tc	Insulated
1-SI-89	6-inch SI to Tc	Insulated
Penetration Area – Elevation 241 Feet Annulus		
1-SI-201	3-inch, Pen. 114	Not Insulated
1-SI-90	3-inch, Pen. 113	Not Insulated
1-SI-195	6-inch, Elev 246 feet	Not Insulated
1-SI-199	6-inch, Elev 248 feet	Not Insulated
1-SI-197	6-inch, Elev 250 feet	Not Insulated
1-SI-206	6-inch, Elev 260 feet Pen. 60	Not Insulated
1-SI-207	6-inch, Elev 258 feet Pen. 61	Not Insulated
Loop Room "A" – Elevation 241 Feet		
1-SI-99	6-inch, off Th Elev 253 feet	Insulated
1-SI-209	6-inch, off 2nd Th Elev 253 feet	Not Insulated
1-RH-MOV-1700	14-inch MOV below Th	Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
Loop Room "B" – Elevation 241 Feet		
1-SI-95	6-inch, off Th	Insulated

	Elev 253 feet	
1-SI-211	6-inch, off 2nd Th Elev 253 feet	Not Insulated
Loop Room "C" – Elevation 241 Feet		
1-SI-103	6-inch, off Th Elev 253 feet	Insulated
1-SI-213	6-inch, off 2nd Th Elev 253 feet	Not Insulated
Containment Basement – Elevation 216 Feet		
1-RH-MOV-1720A	10-inch, Overhead by "B" Accumulator	Insulated
1-RH-MOV-1720B	10-inch, Overhead by "C" Accumulator	Insulated
1-RH-MOV-1701	14-inch, Overhead by RHR Flat	Insulated
1-SI-125	12-inch, beside "A" Accumulator	Not Insulated
1-SI-142	12-inch, beside "B" Accumulator	Not Insulated
1-SI-159	12-inch, beside "C" Accumulator	Not Insulated
Safeguards Building		
1-SI-MOV-1890D	10-inch, top floor	Not Insulated
1-SI-MOV-1890A	10-inch, 1st level	Not Insulated
1-SI-MOV-1890B	10-inch, 1st level	Not Insulated
1-SI-MOV-1890C	10-inch, 1st level	Not Insulated

II. Visual Inspection of ASME Class 2 Pressure Boundary Components (Inside Reactor Containment)

Component Inspected	Location/Description	Insulated/Not Insulated
Penetration Area – Elevation 248 Feet		
1-CH-HCV-1307	Penetration 19	Not Insulated
1-CH-MOV-1380	Penetration 19	Not Insulated
1-CH-TV-1204A	Penetration 28	Not Insulated
RHR Flat – Elevation 231 Feet		
1-RH-P-1B	RHR Flat	Not Insulated
1-RH-7	1-RH-P-1B Outlet Check Valve	Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
1-RH-8	1-RH-P-1B Outlet Isolation Valve	Insulated

1-RH-P-1A	RHR Flat	Not Insulated
1-RH-15	1-RH-P-1A Outlet Check Valve	Insulated
1-RH-16	1-RH-P-1A Outlet Isolation Valve	Insulated
1-RH-19	1-RH-E-1A Inlet Isolation Valve	Insulated
1-RH-E-1A	RHR Flat	Insulated
1-RH-24	1-RH-E-1A Outlet Isolation Valve	Insulated
1-RH-25	1-RH-E-1B Inlet Isolation Valve	Insulated
1-RH-E-1B	RHR Flat	Insulated
1-RH-30	1-RH-E-1B Outlet Isolation Valve	Insulated
1-RH-FCV-1605	RHR Hx Bypass FCV	Insulated
1-RH-31	RHR to Letdown Isolation Valve	Not Insulated
Containment Basement - Elevation 216 Feet		
1-RH-48	3-inch, RHR Recirc Overhead by RHR Flat	Not Insulated
1-RH-1	14-inch 1-RH-P-1B Suction Overhead by RHR	Not Insulated
1-RH-RV-1721B	3-inch, Overhead by RHR Flat	Not Insulated
1-RH-9	14-inch, 1-RH-P-1A Suction Overhead by RHR	Not Insulated
1-RH-RV-1721A	3-inch, Overhead by RHR Flat	Not Insulated
1-RH-HCV-1758	12-inch RHR Hx Return - Common Line	Not Insulated
1-RH-FE-1605	12-inch RHR Common Return	Not Insulated
1-RH-36	6-inch RHR to Pen. 24	Insulated
1-RH-34	6-inch RHR to RP	Insulated
1-CH-HCV-1142	RHR to Letdown HCV	Insulated
1-CH-RV-1203	Letdown RV	Not Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
1-CH-HCV-1310	Normal Charging HCV	Insulated
1-CH-E-4	Excess Letdown Heat Exchanger	Insulated
1-CH-HCV-1137	Excess Letdown	Not Insulated

1-CH-HCV-1389	Excess Letdown Platform	Not Insulated
1-CH-FE-1156	Annulus Near A Loop	Not Insulated
1-CH-FE-1156B	Annulus Near A Loop	Not Insulated
1-CH-HCV-1303A	Annulus Near A Loop	Not Insulated
1-CH-FE-1155	Annulus Near B Loop	Insulated
1-CH-FE-1155B	Annulus Near B Loop	Insulated
1-CH-HCV-1303B	Annulus Near B Loop	Insulated
1-CH-FE-1154	Annulus Near C Loop	Not Insulated
1-CH-FE-1154B	Annulus Near C Loop	Not Insulated
1-CH-HCV-1303C	Annulus Near C Loop	Insulated
1-CH-RV-1382A	Annulus Near Letdown HCVs	Not Insulated

III. Visual Inspection of ASME XI Class 2 and 3 Pressure Boundary Components (Outside Reactor Containment)

Component Inspected	Location/Description	Insulated/Not Insulated
Casing Cooling Pump House		
1-RS-101 1-RS-TK-1 Recirculation line intake	Casing Cooling pump house	Insulated
1-RS-109 1-RS-TK-1 Recirculation line discharge	Casing Cooling pump house	Insulated
1-RS-112 1-RS-P-3A Inlet Isolation valve	Casing Cooling pump house	Insulated
1-RS-127 1-RS-P-3B Inlet Isolation valve	Casing Cooling pump house	Insulated
Yard 1- RS- TK- 1		
Manway for 1-RS-TK-1	1-RS-TK-1	Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
Yard Near 1- QS- TK- 1 (RWST) – Elevation 274 Feet		
1-QS-FE-104	1 1/ 2- inch QS Pump Recirc to RWST	Insulated
1-QS-60	2- inch SI, RH, CH Return	Insulated

	to RWST	
1-QS-40	6- inch RWST Recirc to Mechanical Chillers	Insulated
1-QS-12	1- QS- P- 1B Suction Isolation at RWST	Insulated
1-QS-1	1- QS- P- 1A Suction Isolation at RWST	Insulated
1-QS-61	6- inch SI, RH, CH Return to RWST	Insulated
1-QS-38	LHSI Suction Isolation at RWST	Insulated
1-QS-24	4- inch QS Pump Recirc to RWST	Not Insulated
1-QS-39	2- inch RWST Recirc to Mechanical Chillers	Insulated
1-QS-34	1- QS- MOV- 102A Outlet Isolation Valve	Insulated
1-QS-37	1- QS- MOV- 102B Outlet Isolation Valve	Insulated
1-QS-LT-100A	Inlet Flange to RWST Level Transmitters	Not Insulated
1-QS-LT-100B	Inlet Flange to RWST Level Transmitters	Not Insulated
1-QS-LT-100C	Inlet Flange to RWST Level Transmitters	Not Insulated
1-QS-LT-100D	Inlet Flange to RWST Level Transmitters	Not Insulated
Auxiliary Building - Elevation 274 Feet		
1-CH-FCV-1113A	Boric Acid Filter to Blender FCV	Insulated
1-CH-FCV-1114B	Reactor Coolant Filter to Blender FCV	Not Insulated
1-CH-FCV-1113B	Blender to Charge Pump Suction FCV	Not Insulated
Component Inspected	**Location/Description**	**Insulated/Not Insulated**
1-CH-MOV-1350	Emergency Boration Isolation MOV	Insulated
1-CH-185	Letdown CV to VCT	Insulated
1-CH-FT-1110	Emergency Boration FT	Insulated
1-CH-FT-1113	Boric Acid Filter to	Insulated

	Blender FT	
Auxiliary Building - Elevation 244 Feet		
1-CH-TV-1204B	Pen. 28 Isolation	Insulated
1-SI-MOV-1867A	BIT Inlet MOV	Not Insulated
1-SI-MOV-1867B	BIT Inlet MOV	Insulated
1-SI-TV-1884C	BAT Pump Recirc to BIT	Insulated
1-SI-TK-2	BIT	Insulated
1-SI-FE-1934	BAT Pump Recirc FE	Insulated
1-SI-TV-1884A	BIT Return to BAST	Insulated
1-SI-TV-1884B	BIT Return to BAST	Insulated
1-SI-RV-1857	BIT Outlet RV	Insulated
1-SI-MOV-1867C	BIT Outlet MOV	Insulated
1-SI-MOV-1867D	BIT Outlet MOV	Insulated

Table 2

Components Inspected for Boric Acid Leakage
Surry Power Station Unit 1 (Unit 2 Similar)

I. Visual Inspection of ASME Class 1 RCS Pressure Boundary Components

Component Inspected	Location/Description	Insulated/Not Insulated
Pressurizer Cubicle		
1-RC-MOV-1535	At column 9, 15 feet off floor, 3" valve upstream of PCV-1456	Insulated
1-RC-MOV-1536	At column 9, 14 feet off floor, 3" valve upstream of PCV-1455C	Insulated
1-RC-PCV-1455C	At column 9, 14 feet off floor, 3" valve down-stream of MOV-1536.	Insulated
1-RC-PCV-1456	At column 9, 14 feet off floor, 3" valve down-stream of PCV-1456	Insulated
1-RC-E-2	Manway	Insulated
1-RC-SV-1551A	Pressurizer safety	Not Insulated
1-RC-SV-1551B	Pressurizer safety	Not Insulated
1-RC-SV-1551C	Pressurizer safety	Not Insulated
18' Elevation, Near Pressurizer (outside "C" RCP Motor Room)		
1-RC-PCV-1455A	Against wall	Insulated
1-RC-PCV-1455B	Against wall	Insulated
"A" RCS Loop Room		
1-RC-E-1A	HL & CL manway	Insulated
1-RC-FE-1480	8' off floor (2" loop bypass FE, down-stream of 1-RC-154)	Not Insulated
1-RC-HCV-1556A	Loop fill HCV, off cold leg	Insulated
1-RC-HCV-1557A	Loop drain HCV, off cold leg	Not Insulated
1-CH-LCV-1460A	Letdown LCV, on floor	Not Insulated
1-CH-LCV-1460B	Letdown LCV, on floor	Not Insulated
1-SI-91	Hot leg SI check valve (first valve off loop)	Insulated
1-RH-MOV-1700	On floor, against wall, "A" pump cubicle	Insulated

Component Inspected	Location/Description	Insulated/Not Insulated
"A" Reactor Coolant Pump Cubicle		
1-RC-MOV-1590	HL loop stop	Insulated
1-RC-MOV-1591	CL loop stop	Insulated
1-SI-79	Cold leg SI check valve (first valve off loop)	Insulated
1-SI-241	Cold leg SI check valve (second valve off loop)	Not Insulated
1-SI-109	"A" Accumulator Discharge Check Valve.	Insulated
1-RC-P-1A	RTD thermowell cap	Insulated
"B" RCS Loop Room		
1-RC-E-1B	HL & CL manway	**Insulated**
1-RC-HCV-1557B	Loop drain HCV, off cold leg	Not Insulated
1-RC-HCV-1556B	Loop fill HCV, off cold leg	Insulated
1-RC-FE-1481	2" loop bypass FE, downstream of 1-RC-84	Not Insulated
1-SI-88	Hot leg SI check valve (first valve off loop)	Insulated
"B" Reactor Coolant Pump Cubicle		
1-RC-MOV-1592	HL loop stop	Insulated
1-RC-MOV-1593	CL loop stop	Insulated
1-SI-130	7' off floor against wall	Insulated
1-CH-312	3" Charging check valve next to 1-SI-130 (3" line ties into cold leg down-stream of loop stop)	Insulated
1-CH-430	3" check valve down-stream of 1-CH-312	Insulated
1-SI-82	Cold leg SI check (first valve off loop)	Insulated
1-SI-242	Cold leg SI check valve (second valve off loop)	Not Insulated
1-RC-P-1B	RTD thermowell plug	Insulated
"C" Loop Room		
1-RC-E-1C	HL & CL manways	Insulated
1-RC-FE-1482	2" loop bypass FE	Not Insulated
1-RC-HCV-1556C	Loop fill HCV, off cold leg	Insulated
1-RC-HCV-1557C	Loop drain HCV, off cold leg	Not Insulated

Component Inspected	Location/Description	Insulated/Not Insulated
1-SI-94	Hot leg SI check valve (first valve off loop)	Insulated
"C" Reactor Coolant Pump Cubicle		
1-RC-MOV-1594	HL loop stop	Insulated
1-RC-MOV-1595	CL loop stop	Insulated
1-SI-147	5' off floor, against wall	Insulated
1-SI-85	Cold leg SI check valve (first valve off loop)	Insulated
1-SI-243	Cold leg SI check valve (second valve off loop)	Not Insulated
1-RC-P-1C	RTD thermowell cap	Insulated
Basement		
1-CH-HCV-1311	Inside crane wall, behind 1-RS-E-1C, 3 ft. off floor	Insulated

II. Visual Inspection of ASME Class 2 Pressure Boundary Components (Inside Reactor Containment)

Component Inspected	Location/Description	Insulated/Not Insulated
Penetration Area, 3'6" (Class 2)		
1-CH-455	Near crane wall, inside handrail	Near crane wall, inside handrail Not Insulated
1-CH-TV-1204A		Not Insulated
Regenerative HX Room (Class 2)		
1-CH-HCV-1200A		Insulated
1-CH-HCV-1200B		Insulated
1-CH-HCV-1200C		Insulated
1-CH-RV-1203		Insulated